NON-PUBLIC?: N
ACCESSION #: 9309170264
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Fermi 2 PAGE: 1 OF 8

DOCKET NUMBER: 05000341

TITLE: Reactor Automatic Shutdown Following a False High Water
Level Trip of the Main Turbine Generator
EVENT DATE: 08/13/93 LER #: 93-010-00 REPORT DATE: 09/13/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 93%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: John A. Tibai, Compliance Engineer TELEPHONE: (313) 586-4289

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On August 13, 1993, while performing routine reactor building rounds, a Nuclear Power Plant Operator (NPPO) removed some tape from the valve handle on a manifold to instrument B21-R004B "Reactor Vessel Division 2 Pressure Indicator". While removing the tape residue, the NPPO noticed a pressure spike on instrument B21-R004B and heard the scram pilot valves vent. He called the main control room and was informed that an automatic reactor shutdown had occurred.

An investigation determined the cause of the event to be personnel error. The NPPO inadvertently unseated an instrument calibration/vent isolation valve while removing the tape residue from the valve handle. This induced a pressure transient into an instrument header shared by other instruments. A false high reactor water level signal caused the main turbine generator to trip off. Closure of the main turbine generator control valves initiated a reactor scram.

A lessons learned meeting was conducted. Signs have been posted on instrument racks H21-P004/P005 warning of the sensitivity of these instruments. Lessons learned will be incorporated into the licensed and non-licensed operator training program.

END OF ABSTRACT

"Required Number of Digits/Characters for Each Block"

Initial Plant Conditions:

Operational Condition: 1 - Power Operation
Reactor Power: 93.5 Percent
Reactor Pressure: 1020 psig
Reactor Temperature: 535 degrees Fahrenheit

Description of Event:

On August 13, 1993, at 0946 hours while performing routine reactor building rounds, a Nuclear Power Plant Operator (utility non-licensed)(NPPO)! noticed some tape on a valve handle on a manifold to instrument PI! B21-R004B "Reactor Vessel Division 2 Pressure Indicator". This instrument is located on reactor vessel level and pressure instrument rack RK! H21-P005. To maintain good plant housekeeping the NPPO removed the tape. While the NPPO was removing the residual adhesive material he noticed a pressure spike on instrument B21-R004B and heard the hydraulic control unit HCU! scram pilot valves V! vent. He called the main control room and was informed that an automatic reactor shutdown had occurred.

Subsequent investigation determined that while the NPPO was removing the tape and residual adhesive material from the valve handle, the valve was inadvertently unseated. This induced a pressure transient into a common header shared by other instruments. The NPPO was not aware that the valve he was working on was hydraulically connected through a common instrument reference leg to other instruments on this rack. The pressure transient created a false high reactor pressure vessel RPV! water level signal. This false signal caused the main turbine generator TA! and both reactor feed pumps SK! to trip off line. These steam driven components trip when reactor pressure vessel high water level is indicated to protect the associated turbines from any potential water impingement.

An automatic reactor protection system (RPS)(JC)! shutdown was initiated by the closure of the main turbine generator control valves CV!. All control rods fully inserted AA!. Without the reactor feed pumps in service, reactor pressure vessel water level decreased.

At reactor pressure vessel low water level 3 the following group isolations occurred as expected: 4 JM!, residual heat removal shutdown cooling; 13,

drywell sumps and 15 AC!, traversing in-core probe system. The standby feedwater (SBFW) system was manually started to supply water to the reactor pressure vessel. Reactor pressure vessel water level continued to decrease.

At reactor pressure vessel low water level 2 the following group isolations occurred as expected: 2 SMV!, reactor water sample system; 10 and 11 CE!, reactor water cleanup system (RWCU); 12, torus water management system; 14, drywell and suppression pool ventilation system; 16, nitrogen inerting system; 17, reactor recirculation pump system and primary containment radiation monitoring system and 18 LK!, primary containment pneumatic supply system.

At reactor pressure vessel low water level 2, both the high pressure coolant InJection (BJ)(HPCI)! and the reactor core isolation cooling (BN)(RCIC)! systems automatically started and injected into the reactor pressure vessel to restore reactor water level. Both systems performed as expected. Other events that occurred at reactor pressure vessel low water level 2 were the alternate rod insertion reactor recirculation pump (AD)(RRS)! trip, the control air LD! compressors automatically started, the standby ps treatment system BH! automatically started, the control center ventilation VI! system automatically shifted to recirculation mode, secondary containment VA! isolated, and the low pressure coolant injection (BO)(LPCI)! loop select logic permissive was enabled.

The Nuclear Shift Supervisor (utility licensed)(NSS)! declared an Unusual Event due to HPCI injecting into the reactor vessel.

While recovering from the automatic reactor shutdown, main condenser SD! vacuum was lost resulting in a group 1 isolation. This isolation group includes the main steam isolation valves (MSIVs). C
ndenser vacuum was
lost because the turbine gland sealing system TC! supply steam from the main steam manifold was isolated in accordance with plant procedures because the system's current instrumentation cannot respond to the

different modes of system operation and operator action was required to maintain the system in service. Without the turbine gland sealing system supply steam, sealing steam was not provided to the turbine. Without the steam seals, air in-leakage to the main condenser occurred.

Following closure of the MSIVs, reactor pressure was controlled by manual operation of the safety relief valves. Both divisions of the residual heat removal system were placed in the suppression pool cooling mode. Reactor level was controlled by the standby feedwater and condensate systems.

To mitigate any coolant thermal stratification, the operators determined the need to restart the reactor recirculation system (RRS). To start the RRS, Technical Specification 4.4.1.1.4.a requires that the temperature differential between the reactor coolant within the RPV dome and the bottom head drain be less than 145 degrees F. To obtain an accurate bottom head drain temperature, the RWCU system was restored to service. RWCU was lined up to take suction from both the RRS loops and the bottom head drain. The initial bottom head drain temperature log entry following the restoration of the RWCU system was 290 degrees F. Using this temperature reading, the reactor cooldown rate was calculated to be 101.3 degrees F/hr. This exceeded the Technical Specification limit of 100 degrees F/hr. Cooldown rates calculated for the same time period from the RPV shell temperature and the RPV dome pressure were well below the Technical Specification cooldown limits.

RWCU suction was lined up to both the bottom head drain and the RRS loops. The higher percentage of flow was from the RRS loops. The RWCU flow through the bottom head was only slightly greater than the cooling water input to the RPV bottom head region from the control rod drive AA! system. To increase the bottom head drain temperature, flow through the bottom head drain was increased. This was accomplished by adjusting RWCU throttle valve G33-F102 which reduced the input to RWCU suction from the RRS loops. During this evolution the RPV bottom head drain line temperature increased 135 degrees F in one hour. During this same time period, temperature data from the RPV shell and RPV dome indicated a slight temperature decrease.

An engineering evaluation to determine the effects of the out-of-limit temperature cooldown and heatup conditions on the structural integrity of the reactor coolant system was performed as required by the Technical Specification action statement as described below.

During a post event review, the control room operators noted that during the event, an unusually large number of control panel indicating bulbs burned out. The operators found it necessary to either replace the bulbs or use alternate indications to verify actions taken. Although a nuisance, at no time did this cause the operators to take inappropriate actions.

Some other distractions noted by the operators during this event were that there was no immediate indication of an isolation for groups 14 and 16 on the isolation mimic, the division 1 torus and drywell pressure recorder PR! displayed anomalous readings, the power supply to BOP battery charger EI! 2PC3-16 tripped, the SRV open alarm did not display on the sequence of events recorder IQ!, and RHR riser differential pressure signal conditioner PDM! B31-N610B failed. At no time did these problems cause the operators to take inappropriate actions. Deviation event reports (DER) were issued for each of these in accordance with the corrective action program. A review and analysis will be conducted and appropriate corrective actions will be taken.

Cause of Event

The cause of this event is personnel error. The NPPO inadvertently unseated an instrument calibration/vent isolation valve VTV! to instrument B21-R004B, "Reactor Vessel Division 2 Pressure Indicator," located on instrument rack H21-P005 while removing tape residue from the valve handle. This instrument shares a common reference leg with other reactor vessel level and pressure instruments. When the calibration/vent isolation valve was unseated a pressure transient was induced into the common reference leg when the volume between the calibration/vent isolation valve and the volume chamber isolation valve pressurized. The reference leg experienced a brief depressurization which caused reactor level transmitters LT! B21-N095B and B21-N095D to indicate a high reactor water level. The false high reactor water level signal caused the main turbine generator to trip off. Closure of the main turbine generator control valves initiated a reactor scram. Control room alarms ALM! and plant equipment responses were consistent with the expected response of instruments that share the affected reference leg.

The cause of the group 1 isolation was a loss of condenser vacuum. The turbine gland sealing system supply steam from the main steam manifold was isolated at the time the reactor automatic shutdown occurred. During operation at normal loads the turbine becomes self-sealing as the seal header is supplied with steam from the high pressure turbine center gland. After the reactor automatic shutdown the steam supply from the main steam manifold was not opened to provide seal steam pressure. The turbine sealing was lost and condenser vacuum decreased due to air in-leakage. Although the system is intended to operate automatically, the system's current instrumentation cannot respond to the different modes of system operation. A new design had been prepared and had been scheduled for Implementation during the next refueling outage.

Analysis of Event:

A false high reactor water level signal caused the main turbine generator and both reactor feed pumps to trip off. Closure of the main turbine generator control valves initiated a reactor scram. Actual reactor water level was approximately 196 inches, within the normal operating band. All safety systems functioned as designed.

The transient was compared with a turbine generator trip analysis described in chapter 15 of the UFSAR. In the analyzed transient, peak pressure increase would be about 190 psi. The actual peak pressure increase was 50 psi during this event, well within the analyzed event. The reactor pressure did not increase to the point where the vessel safety relief valves would lift automatically. Following the group 1 isolation on low condenser vacuum, due to a loss of turbine sealing, the SRVs were manually actuated to control reactor pressure.

The minimum reactor water level was approximately 105 inches. The decrease in water level was less severe than a typical loss of feedwater event because the automatic reactor shutdown occurred when the reactor water level was in the normal operating range and standby feedwater was manually started immediately following the reactor shutdown. In a loss of feedwater event an automatic reactor shutdown would be initiated on a low reactor water level signal, approximately 25 inches below the normal operating range. During the November 1992 loss of feedwater event described in Licensing Event Report 92-012 the minimum reactor water level was about 77 inches.

During this event the Technical Specification RPV cooldown and heatup rate limits of less than 100 degrees F/hour were exceeded. The maximum calculated cooldown rate was 101.3 degrees F/hour. The maximum calculated heatup rate was 135 degrees F/hour. Both the cooldown and heatup rates were calculated using temperature readings from the reactor bottom head drain line. Similar calculations using temperature data from the RPV shell and the RPV dome showed the cooldown rate to be well below the Technical Specification limit, further, this data did not show any heatup. However, in accordance with the Technical Specifications, an engineering evaluation of the calculated high cooldown and heatup rates was performed. General Electric had previously analyzed heatup and cooldown rates as high as 160 degrees F/hour for one hour and 130 degrees F/hour for two hours and concluded that the reactor coolant system remained acceptable for continued operation following such thermal events.

Corrective Actions:

HPCI and RCIC injection lines were inspected for potential pipe support damage. No damage was found.

The main steamline and bypass lines were inspected for potential pipe support damage. No damage was found.

Signs have been installed on reactor pressure and level instrument racks H21-P004 and H21-P005 warning of the sensitivity of these instruments to inadvertent actuations.

Engineering Design Package EDP-10257, "Gland Steam Controller Changes for Gain Differences of Valving/Service" will be implemented during the next refueling outage. This EDP will allow the main turbine gland sealing system to work automatically. This will free the control room operators from having to manually control this system.

Abnormal Operating Procedure 20.000.21, "Reactor Scram," was revised to include restoration of the gland sealing system. This administrative control will be necessary until EDP-10257 is implemented.

Licensed Operator Training will:

Include lessons learned from this event.

Emphasize the importance of restoring the gland sealing system immediately following a turbine trip.

This training will be completed by October 28, 1993.

Non-licensed operator training will include lessons learned from this event. This training will be complete by October 28, 1993.

Other automatic systems which require manual control during transients will be reviewed to assure that there are appropriate compensatory measures such as training and/or procedures in place and that long term corrective actions are developed. This would include a review of any related pending modifications. This review will be completed by November 1, 1993.

Other deficiencies noted in the last two paragraphs of the "description of events" above have been documented on deviation event reports (DERs). In accordance with the corrective action program a review and analysis will be conducted and appropriate corrective actions will be taken.

Previous Similar Events:

None

ATTACHMENT 1 TO 9309170264 PAGE 1 OF 1

Douglas R. Gipson
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Nuclear Generation

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10CFR50.73

September 13, 1993
NRC-93-0092

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Licensee Event Report (LER) No. 93-010

Please find enclosed LER No. 93-010, dated September 13, 1993 for a reportable event that occurred on August 13, 1993. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Mr. John A. Tibai, Compliance

Engineer, at (313) 586-4289.

Sincerely,

Enclosure: NRC Forms 366, 366A

cc: T. G. Colburn
W. J. Kropp
J. B. Martin
M. P. Phillips
P. L. Torpey

Wayne County Emergency
Management Division

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